Comprehensive Risk Metrics for Nuclear Reactor Regulation

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Executive Summary

The U.S. Nuclear Regulatory Commission (NRC) is in the midst of a rulemaking to develop a new technology-inclusive, risk-informed and performance-based (TI-RIPB) regulatory framework in 10 CFR Part 53. A key question for NRC under this rulemaking is how to effectively and efficiently meet the NRC legal standard for “reasonable assurance of adequate protection” without the use of the existing amalgam of regulatory requirements on siting, design, analysis, construction, maintenance, and operation that have enabled decades of safe operation of nuclear power plants in the United States.

Comprehensive risk metrics and risk surrogates have been used by applicants, licensees, and the NRC over the past several decades as a factor that can inform regulatory decisions but not as a formal regulatory requirement. The proposed Part 53 rule now seeks to codify a licensing requirement for applicants to develop, propose, and demonstrate compliance with a comprehensive risk metric or risk surrogate. While this requirement is a departure from previous uses of risk metrics, use of a comprehensive risk metric or risk surrogate could enable the development of a complete TI-RIPB regulatory framework for advanced reactors.

This paper describes progress by the NRC staff to develop the proposed 10 CFR Part 53 regulatory framework, how the NRC staff and the Commission have incorporated the concept of risk metrics in the proposed rule, the history of risk metrics used by the NRC as a part of licensing activities, the proposed use of a comprehensive risk metric to support technology-inclusive, risk-informed and performance-based (TI-RIPB) regulation in Part 53, and key considerations for the development and implementation of comprehensive risk metric requirements and guidance for NRC staff and prospective applicants. The derivations for the numerical values for the existing NRC’s risk metrics (quantitative health objectives and light water reactor surrogate risk objectives) are provided in Appendix A of this paper.

This paper seeks to build consensus on the basis for risk metrics to help inform on-going discussions between NRC staff, industry, and other stakeholders on the implementation of comprehensive risk metrics in the proposed 10 CFR Part 53 regulatory framework. The NRC, industry, and other stakeholders should collaborate on the development of specific comprehensive risk metrics through workshops and other public forums to ensure that future risk metrics are usable by applicants, reviewable by NRC, implementable by operators, and contribute to a complete TI-RIPB licensing basis for new nuclear reactors.

Development and use of comprehensive risk metric alone will not provide “reasonable assurance of adequate protection” but can contribute to an overall finding of “reasonable assurance of adequate protection” when assessed in combination with other safety limits and licensing requirements such as chronic and acute dose limits for members of the public, worker dose limits, environmental effluent limits, and other operational program requirements in future TI-RIPB regulatory frameworks.

Table of Contents

1.0. Introduction

The U.S. Nuclear Regulatory Commission's (NRC's) legal standard for licensing and regulatory decisions is defined as "reasonable assurance of adequate protection – not the elimination of all risk."[1] Historically, this legal standard for licensing decisions has been met, in part, by ensuring compliance with prescriptive, deterministic, and technology-specific regulations[2]. Decades of experience with the design, regulation, and operation of large light water reactors have resulted in an amalgam of regulatory requirements on siting, design, analysis, construction, maintenance, and operation. Compliance with this complete set of requirements has resulted in nuclear power plants that operate safely and can meet societal requirements for the continued use of nuclear energy. Ensuring compliance with these prescriptive, deterministic, and technology-specific regulatory requirements has enabled the NRC to meet their legal standard of "reasonable assurance of adequate protection" for nuclear energy.

Many of the existing prescriptive, deterministic, and technology-specific regulations, however, may no longer be applicable, appropriate, or sufficient as the NRC begins to license new and novel nuclear technologies. New reactor license applicants are making significant changes in the siting, design, analysis, construction, maintenance, and operation of nuclear reactors. The historic regulatory and operational experience with large light water reactors may have more limited applicability to the licensing and operation of advanced reactors. Effectively and efficiently meeting the NRC legal standard for "reasonable assurance of adequate protection" will require alternative approaches for licensing and regulatory decision-making for new nuclear technologies.

Historically, the NRC staff have evaluated new and novel nuclear technologies on an application-specific basis. Initial licensing of nuclear reactors by the Atomic Energy Commission (AEC), the regulatory predecessor to the NRC, was completed using a "licensing by order" approach. "Licensing by order" is characterized by a regulator making licensing decisions on a case-by-case basis, largely guided by agency staff judgement and assessment of an individual applicant's compliance with high-level qualitative regulatory

[1] Memo to New Reactor Business Line from F. Brown, NRO re: Expectations for New Reactor Reviews

[2] Prescriptive regulations specify the methods an applicant must use to demonstrate compliance with regulatory requirements. Prescriptive regulation can be contrasted with performance-based regulation that specifies the measurable outcome that must be met by an applicant regardless of methods used. Deterministic regulations specify the use of methods that assess safety without consideration for event probability or frequency. Deterministic regulations can be contrasted with risk-informed regulations that include evaluations of event probability and consequence for prioritization and assessment of event risks. Technology-specific regulations specify regulatory requirements that are applicable to a single reactor technology and not generally applicable to all technologies. Technology-specific regulations can be contrasted with technology-inclusive regulations that are broadly applicable to any reactor technology.

requirements. The “licensing by order” approach is highly flexible for applicants and is easy for the regulator to initially develop (requiring limited resources and staff expertise) but requires significant regulator and licensee resources to complete each review. These reviews can also be unpredictable as agency criteria and standards for licensing decisions could vary significantly on an application-by-application basis. Changing regulatory requirements and standards for reactors that required redesigns or backfits for reactors under construction in the 1970s were cited by the Congressional Budget Office as a major source of delay for new reactor licensing and construction.[3]

As the AEC (and then the NRC) began to codify licensing precedents in agency guidance and regulations, nuclear power plant licensing evolved towards “licensing by rule”. “Licensing by rule” is characterized by a regulator making licensing decisions based on an applicant’s demonstrated compliance with quantitative or qualitative rules for licensing. These rules outline the conditions and requirements for licenses and focus regulator reviews on applicant compliance with specific regulations. This “licensing by rule” approach is more predictable for applicants and can streamline regulator decision making but can be inflexible for applications that meet the intent of the regulatory requirements but do not conform with the existing rules. For applicants that do not meet the specific regulations, the regulator may grant case-by-case exemptions to enable licensing, but this process can be unpredictable as well as time and resource intensive depending on the breadth and depth of the requested exemptions. The process can also become unpredictable if a regulator’s definitions of acceptable deviations and exemptions from the existing regulations changes over time. The Government Accountability Office noted in a 2015 report on new reactor deployment that the use of regulatory exemptions for licensing advanced reactors under the existing regulatory framework would create greater licensing uncertainties than existing reactor designs that do not require exemptions.[4]

New advanced reactor applications are expected to need exemptions or alternative regulatory requirements to the existing prescriptive, deterministic, and technology-specific rules used by the NRC. While it would be possible for the NRC to use a “licensing by order” process to license new nuclear technologies, the process can be ineffective and inefficient. Use of application-specific licensing requirements on a case-by-case can lead to inconsistency, unpredictability, and uncertainty throughout the licensing process. While applicants and staff can work successfully to complete a new reactor licensing process that

[3] Delays in Nuclear Reactor Licensing and Construction: The Possibilities for Reform
[4] Technology Assessment: Nuclear Reactors: Status and Challenges in Development and Deployment of New Commercial Concepts

meets the legal standard of "reasonable assurance of adequate protection,"[5] the process may be inefficient especially for large numbers of applications. Creation of a performance-based, risk-informed, and technology inclusive regulatory framework and rules for the NRC will enable the effective and predictable "licensing by rule" of advanced nuclear reactors.

2.0. Development of technology-inclusive, risk-informed, and performance-based (TI-RIPB) regulation

Congress directed the NRC in 2019 to develop and implement a new technology-inclusive, risk-informed, and performance-based (TI-RIPB) regulatory framework to create a more effective approach to licensing advanced nuclear reactors.[6] This direction by Congress built on decades of work by the NRC and industry to create more effective and efficient regulation for nuclear reactors using TI-RIPB processes and rules.[7]

Creation of the new regulatory framework would enable effective "licensing by rule" for all reactor technologies and reducing reliance on licensing exemptions without reverting to "licensing by order" for new nuclear technologies that do not align with existing prescriptive, deterministic, and technology-specific regulations. A technology-inclusive framework ensures that the regulatory framework could be applied to any nuclear reactor license application and would not be limited to a specific reactor design or technology. Implementing risk-informed decision-making approaches would enable applicants and regulators to focus on the aspects of design, construction, and operation that are most significant to safety. Use of performance-based regulatory requirements would help focus on the safety outcomes of reactor siting, design, analysis, construction, maintenance, and operation and not prescribe the specific actions used to ensure safety.

Development of a new TI-RIPB regulatory framework for advanced reactors, however, has been challenging. Effective TI-RIPB regulation must be:

- Flexible: enabling evaluation of any reactor siting, design, analysis, construction, maintenance, and operation
- Predictable: facilitating licensing for different applicants with a clear understanding of regulatory requirements, schedule, and necessary resources

[5] The NRC has successfully worked with Kairos Power and Abilene Christian University to issue construction permits for advanced research and test reactors using regulatory exemptions within the existing regulatory framework.
[6] [SECY-20-0032: Rulemaking Plan on "Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (RIN-3150-AK31; NRC-2019-0062)"]
[7] [Risk-informed, performance-based safety: Past, present, and future]

- Self-consistent: ensuring the same acceptable level of safety for existing and new nuclear power plants
- Implementable: ensuring that any new regulatory requirements do not create unnecessary burden for applicants or the regulator to demonstrate safety

If a new TI-RIPB regulatory framework fails to provide a flexible, predictable, self-consistent, and implementable licensing process for advanced reactors, it will not result in a more efficient and effective licensing process for advanced reactors.

Previous efforts by the NRC and other stakeholders to create a TI-RIPB regulatory framework have focused on adapting or modifying the existing regulatory framework in 10 CFR Part 50 and 10 CFR Part 52 to be more TI-RIPB. The Licensing Modernization Project (LMP) presented in NEI 18-04, for example, created a TI-RIPB process for identifying and selecting licensing basis events, classifying the safety significance of systems, structures, and components (SSCs), and demonstrating defense-in-depth adequacy.[8] The LMP process, endorsed by NRC staff in NRC Regulatory Guide 1.233, did not supplant all of the regulatory requirements on siting, design, analysis, construction, maintenance, and operation of nuclear reactors. [9] Instead, it helped "inform the licensing basis and content of applications" for non-light water reactors and build upon or provide technical basis for exemption to the other regulatory requirements in 10 CFR Part 50 and 10 CFR Part 52.[10]

The LMP process enabled the use of TI-RIPB methods for certain key activities and helped facilitate more consistent licensing – moving closer to the goal of "licensing by rule" for advanced reactors – but still required applicants to comply with (or seek exemption and propose alternatives for) many of the prescriptive, deterministic, and technology-specific regulations that provided a legal basis for staff determinations of "reasonable assurance of adequate protection." Several new reactor applicants have expressed an interest in using the LMP methodology and Reg. Guide 1.233 to support licensing non-light water reactors using 10 CFR Part 50.[11,12] Implementation of the LMP process does not result in a complete TI-RIPB regulatory framework and still relies on the existing regulatory requirements and licensing

[8] NEI 18-04, Rev. 1, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactors"
[9] Regulatory Guide 1.233, Revision 0, Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors
[10] Draft Interim Staff Guidance - DANU-ISG-2022-01, Advanced Reactor Content of Application Project, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications - Roadmap"
[11] Regulatory Engagement Plan for the Natrium™ Reactor
[12] Risk-Informed Performance-Based Licensing Basis: X-Energy's Approach to NEI 18-04 Implementation

basis assumptions related to siting, design, analysis, construction, maintenance, and operation to provide "reasonable assurance of adequate protection."

3.0. NRC's proposed use of risk metrics in 10 CFR Part 53

The current development activities for a TI-RIPB regulatory framework for advanced reactors began in 2019 with the Nuclear Energy Innovation and Modernization Act (NEIMA). NEIMA directed the NRC to "complete a rulemaking to establish a technology-inclusive regulatory framework for optional use by commercial advanced nuclear reactor applicants for new reactor license applications." [13] This rulemaking activity is commonly referred to as "Part 53" based on its expected location in Title 10 of the Code of Federal Regulation (i.e., 10 CFR Part 53). The NRC staff began development of a rulemaking plan for Part 53 in 2019 and described plans to leverage prior work on the LMP process as the basis for Part 53.[14] One area of focus during the Part 53 rulemaking process has been the role of risk metrics in the new TI-RIPB regulatory framework.

Risk metrics are "quantitative measures of risk to the public from reactor operations up to and including severe core damage accidents" and help to quantify the overall risk associated with plant operations.[15] The foundational risk metrics for the NRC were outlined in the 1986 Safety Goal Policy Statement which detailed a new Quantitative Health Objective (QHOs) for nuclear power plant operation.[16] Specific applicant and NRC staff usage of risk metrics (including QHOs and surrogate metrics such as Core Damage Frequency and Large Early Release Frequency) have evolved over time, but quantitative risk metrics have consistently been used to inform regulatory decisions and not used as regulatory requirements. While the LMP process had applicants evaluate plant risk against the QHOs, they were not used as an explicit regulatory requirement and were instead used to help classify the safety significance of different plant systems, structures, and components (SSCs) and licensing basis events (LBEs).[17]

Preliminary NRC staff drafts of the proposed Part 53 rule as early as 2020 indicated that NRC staff intended to codify the use of the QHOs as a risk metric for advanced reactors. Early drafts of the proposed rule would have, for the first time, required applicants to demonstrate that a nuclear power plant would explicitly satisfy the QHOs. Applicant demonstration of

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[13] Text - S.512 - 115th Congress (2017-2018): An act to modernize the regulation of nuclear energy

[14] SECY-20-0032: Rulemaking Plan on "Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (RIN-3150-AK31; NRC-2019-0062)"

[15] White Paper on Options for Risk Metrics for New Reactors.

[16] Safety Goals for the Operation of Nuclear Power Plants: Policy Statement

[17] NEI 18-04, Rev. 1, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactors"

compliance with the QHOs, however, would effectively require applicants to develop a detailed probabilistic risk assessment (PRA) that quantified plant safety. While this requirement was technology-neutral and risk informed, it was not fully performance-based or technology-inclusive. Demonstration of compliance with the QHO risk metrics was a prescriptive regulatory requirement that limited the use of some analysis and evaluation methods that could otherwise be used to demonstrate compliance with performance-based requirements. Thus, while codification of the LMP and QHOs would benefit some advanced reactor applicants, it did not create a complete TI-RIPB regulatory framework that was flexible, predictable, self-consistent, and implementable.

During the Part 53 development process, external stakeholders provided feedback to the NRC staff on alternative rule language and frameworks that could enable development of a complete TI-RIPB regulatory framework. Both NIA and the Nuclear Energy Institute (NEI) submitted public comments to the NRC that outlined more complete TI-RIPB regulatory frameworks that would be more flexible, predictable, self-consistent, and implementable than the regulatory framework proposed by staff.[18,19] These comments described how the QHOs and other prescriptive requirements could be replaced by a more performance-based regulatory framework that enabled TI-RIPB licensing of new reactors. The NRC staff did not incorporate these comments, and the draft proposed Part 53 rule submitted to the Commission in March 2023 for Commission approval still included demonstrated compliance with QHOs as a licensing requirement.[20]

Commission votes on the draft proposed Part 53 rejected the staff's inclusion of QHOs as a licensing requirement and provided new direction to the staff on the use of risk metrics. Each of the four Commissioners' votes (as well as the final direction to staff in the Staff Requirements Memo) detailed specific feedback on the use of risk metrics:

- Commissioner Wright voted to replace the QHO requirement with a new applicant-defined plant risk metric (or set of risk metrics) to help assess plant safety. The vote noted that the metric could include the QHOs, existing light water reactor risk metrics such as core damage frequency (CDF), large release frequency (LRF), large early release frequency (LERF), or new risk metrics developed and justified by the applicant.[21]

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[18] NRC-2019-0062-0162 Comment (080) from Patrick White on behalf of the Nuclear Innovation Alliance FR Doc # 2020-24387 (ML21321A284)
[19] NRC-2019-0062-0045 Comment (16) from Marcus Nichol on behalf of Nuclear Energy Institute on PR-53 – Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (ML21042B889)
[20] SECY-23-0021: Proposed Rule: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (RIN 3150-AK31)
[21] VR-SECY-23-0021: Proposed Rule: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (RIN 3150-AK31) (Wright)

- Commissioner Caputo voted to fully remove the QHOs from the draft rule and instead keep QHOs as "a gauge of safety significance in our regulatory and licensing processes". The vote cited prior NRC decisions and recommendations (including by the Commission, NRC management and staff, and the Advisory Committee on Reactor Safeguards) to keep QHOs as a factor that can be used to assess safety but not as a formal decision criterion, as well as the challenges of implementing the QHOs for advanced reactors.[22]
- Commissioner Crowell voted to replace the QHOs with a more generic risk metric for advanced reactors in Part 53 that would enable more flexible and predictable regulation. The vote noted that "Parts 50 and 52 do not need a codified cumulative risk measure because those frameworks instead rely on a system of deterministic, prescriptive requirements that, taken together, provide the safety case" and argued that establishing a process for applicant-defined approaches and public examination of proposed risk metrics were key to enabling consistent and clear regulation.[23]
- Chair Hanson voted to require applicants to define a "comprehensive risk metric" for new reactors that would ensure an acceptable level or risk from nuclear power plants. The vote discussed the potential impact of QHO codification stating that "[e]nshrining the QHOs as the exclusive metric for establishing a cumulative risk standard in Part 53 may introduce an unnecessarily inflexible requirement where a broader standard could be used to meet the same need". [24]

Though the individual Commission's votes differed, there was unanimous agreement that codification of the QHOs would not result in a complete TI-RIPB regulatory framework and could create a less flexible and less implementable regulatory framework for advanced reactors.

The Commission released the Staff Requirements Memo (SRM) on the proposed draft Part 53 rule on March 4, 2024. The SRM directed the staff to make significant changes to the Part 53 rule text with the goal and specifically disapproved codification of the QHOs in Part 53. Instead, the Commission directed staff to revise the draft proposed Part 53 rule to include a new requirement for a "comprehensive plant risk metric". This new risk metric was intended to "approximate the total overall risk from the facility" and be "part of a suite of

[22] VR-SECY-23-0021: Proposed Rule: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (RIN 3150-AK31) (Caputo)
[23] VR-SECY-23-0021: Proposed Rule: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (RIN 3150-AK31) (Crowell)
[24] VR-SECY-23-0021: Proposed Rule: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (RIN 3150-AK31) (Hanson)

regulatory requirements that when considered holistically, form the basis for the NRC's decision making".[25] The Commission direction on the "comprehensive risk metric" was intended to refocus Part 53 as a more complete TI-RIPB regulatory framework for advanced reactors.

The NRC staff completed revisions to the draft proposed Part 53 rule based on the Commission direction in the SRM and the proposed rule was released for public comment on October 31, 2024. The NRC staff removed the requirement for the QHOs per the Commission direction in the SRM and added new requirements for a "comprehensive risk metric" in the proposed rule. The proposed rule text related to plant safety objectives and criteria (10 CFR Part 53.220) states the applicant must demonstrate that:

> "(b) The analysis of risks to public health and safety resulting from [licensing basis events] LBEs other than [design basis accidents] DBAs under §53.450(e) includes comprehensive risk metrics that satisfy associated risk performance objectives that are acceptable to the NRC and provide an appropriate level of safety."[26]

The proposed Part 53 rule preamble text and rule text contain limited additional information regarding the implementation of the new comprehensive risk metrics. Much of the proposed preamble text and rule text directly reflects the Commission direction in the SRM and individual Commissioner votes but does not provide additional detail on the technical or regulatory basis for developing or proposing a comprehensive risk metric. While this reflects NRC staff implementation of the Commission direction, it also creates uncertainty. It is not clear whether NRC staff, management, and the Commission are aligned with the conceptual and technical basis that industry and potential applicants may use to develop and propose new comprehensive risk metrics.

This paper seeks to build consensus on the basis for risk metrics to help inform on-going discussions between NRC staff, industry, and other stakeholders on the implementation of comprehensive risk metrics in the proposed 10 CFR Part 53 regulatory framework. First, the conceptual basis for risk metrics within a TI-RIPB regulatory framework are presented and the historical definitions and usages of risk metrics are detailed. Second, the use of a comprehensive risk metric for the proposed Part 53 rule is discussed and additional discussion on key factors affecting the development and proposal of a risk metric are presented. Finally, implementation considerations for the use of comprehensive risk metrics within the proposed Part 53 rule and highlighted. Appendix A provides derivations for the

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[25] SRM-SECY-23-0021: Proposed Rule: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (RIN 3150-AK31)
[26] Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors: Proposed Rule

numerical values for the existing NRC's risk metrics – the quantitative health objectives (QHOs) and light water reactor surrogate risk objectives.

Development, proposal, acceptance, and implementation of comprehensive risk metrics for advanced reactors to support a complete TI-RIPB regulatory framework in Part 53 will be critical to ensuring that the new regulatory framework can enable the effective, efficient, and predictable licensing of advanced reactors.

4.0. Conceptual basis for risk metrics in TI-RIPB regulatory frameworks

Comprehensive risk metrics and quantitative health objectives are, at their core, an imperfect solution to a challenging problem: defining and regulating a societally acceptable exposure to an uncertain, low-probability, high-consequence, externally imposed hazard.

Nuclear power regulation has historically differed from other energy generating technology regulations because the harm and hazards of large light water reactors are uniquely characterized by high-consequence, low-probability hazards. Other energy generating technologies (e.g., fossil fuel power plants) are typically characterized by high-probability, lower-consequence hazards (e.g., continuous emissions and public exposure to particulate and chemical releases) that can be more easily controlled or mitigated on an on-going basis by design and operation. While high-probability, lower-consequence hazards may create the same or higher actual or expected harm on society (e.g., observed and expected deaths attributed from coal power plant operation far exceed those from nuclear power plant operation[27]), public perception of the danger of low-probability, high-consequence hazards results in greater concern in regulating the operational safety of nuclear power plants.[28]

The challenge posed by facilities with high-consequence, low-probability hazards is that it is difficult to conclusively demonstrate their safety or effectively bound the cumulative consequences associated with their long-term operation. Facilities with high-probability, low- or moderate-consequence hazards can be continuously monitored to ensure that the individual or cumulative impact of the facility is socially acceptable. If a facility deviates from acceptable operation through observed or measured harm, then a regulator can make corrections (in real time) to ensure that continued facility operation will satisfy socially acceptable safety limits. Facilities with high-consequence, low-probability events are more challenging to regulate because a single event can result in harm that is not socially

[27] Mortality risk from United States coal electricity generation
[28] Perception of risk posed by extreme events

acceptable. As a result, the regulator must focus on ensuring that the cumulative risk of facility operation (based on the expected probability and consequence of event sequences) does not exceed a societally acceptable level.

The NRC is tasked with ensuring that the operation of nuclear power plants and other activities involving radiation and radioactive material do not exceed a societally acceptable level of consequence and risk. The NRC's statutory mission is to provide "reasonable assurance of adequate protection" of public health and safety, but these terms have not been explicitly defined in legislation or regulation. The opacity of this regulatory standard enables the NRC to manage a variety of different regulatory situations but can also create uncertainty and limit regulatory predictability – especially for novel technologies or use cases. The NRC typically interprets the standard of "reasonable assurance of adequate protection" based on Commission direction, regulatory precedent, and stakeholder input.

To further examine the standard of "reasonable assurance of adequate protection", the standard can be simply separated into two different but related criteria:

- "adequate protection" – regulatory limits based on consequences associated with hazards
- "reasonable assurance" – confidence of meeting regulatory limits

These two criteria are not unique to nuclear regulation and, in fact, implicitly or explicitly underlie regulation for any technology. What has historically made nuclear technology regulation unique, however, is the potential for catastrophic acute event consequences that extend far beyond the plant site boundary and contaminate surrounding areas for decades. Theoretical studies such as the 1957 "WASH-740" report described the theoretical possibilities associated with catastrophic accidents at mid-sized nuclear facilities (500 megawatt thermal light water reactor) and actual nuclear accidents such as the 1986 Chernobyl Nuclear Power Plant disaster demonstrated potential and real impacts of poorly designed, poorly operated, and poorly regulated commercial nuclear power plants.

A single event at even a relatively small nuclear power plant (e.g., hundreds of megawatts) could have physical, social, political, and economic impacts on tens to hundreds of thousands of people living tens or hundreds of miles from the facility.[29] While these catastrophic events may be extremely low probability, few if any other technologies or activities have the potential for such a high-consequence event. Maintaining "reasonable assurance of adequate protection" through regulatory oversight is intended to serve as the societal check on activities that may have significant impacts on the broader population.

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[29] Theoretical Possibilities And Consequences Of Major Accidents In Large Nuclear Power Plants (WASH-740)

The potential for catastrophic event consequences from nuclear power plant operation creates a unique challenge related to definition and enforcement of "reasonable assurance" and "adequate protection" for nuclear regulation in the United States. The design and operation of modern large light water reactors (LWRs) results in an extremely low likelihood of catastrophic accidents, but it is impossible to fully eliminate the potential for catastrophic event consequences due to the inherent hazards associated with release pathways for radiological material from large core inventories.

Nuclear regulations in the United States have evolved over the past seven decades to provide "reasonable assurance of adequate protection" for large LWRs using a combination of prescriptive design, analysis, and operational requirements. The development and deployment of new reactors with different design features, reactor power outputs (and radionuclide inventories), and operational characteristics will require the NRC to develop and implement regulatory frameworks that also enable them to provide "reasonable assurance of adequate protection".

The development and use of quantitative performance-based risk metrics are one way to enable more consistent and predictable comparison of the risks and hazards of different activities within a TI-RIPB regulatory framework even when the specific activity or technology risks have differing probabilities and consequences. Compliance with these risk metrics enables applicants to provide regulators with "reasonable assurance of adequate protection" for a wide variety of different technologies with different risk characteristics.

5.0. Historical definitions of risk metrics

The regulatory interpretation of "reasonable assurance of adequate protection" is frequently connected to the 1961 Supreme Court case in which the Court ruled that the AEC must have a finding of "reasonable assurance" that the facility operation will "provide adequate protection" before permitting reactor construction and operation.[30] The AEC and the NRC were both given significant discretion by courts on the regulatory interpretation of "reasonable assurance of adequate protection" but recommendations from the President's Commission on the Accident at Three Mile Island led to the NRC's development of an explicit policy statement on the safety philosophy inherent in the NRC's decision making process.[31]

The Commission published a final policy statement on safety goals in 1986 that established both qualitative and quantitative safety goals to help guide and support agency decision

[30] Power Reactor Development Co. v. Electricians

[31] 1986 Policy Statement on Safety Goals for the Operations of Nuclear Power Plants

making and assess compliance with the NRC goal of "reasonable assurance of adequate protection". The overall qualitative safety goals established by the Commission are[32]:

1. Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
2. Societal risks to life and health from nuclear power plant operations should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

Based on these two qualitative safety goals, the Commission established quantitative risk objectives that could be used when "determining achievement of the qualitative safety goals."[33] The Commission noted in their 1986 Policy Statement on Safety Goals for the Operations of Nuclear Power Plants that "no death attributable to the nuclear power plant operation will ever be 'acceptable' in the sense that the Commission would regard it as a routine or permissible event" but recognized that calculable risk was important to improved regulatory decision making. The Commission's quantitative risk objectives were defined as:[34]

1. The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
2. The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

These quantitative risk objectives created a basis for evaluating the risk of nuclear power plant design and operation. The value of one-tenth of one percent (0.1 percent) was selected by the Commission because it "appropriately reflects both of the qualitative goals – to provide that individuals and society bear no significant additional risk."[35] The Commission, however, also notes that the exceeding the 0.1 percent objective does not create a "significant additional risk" and that the 0.1 percent objective was intended to support the expectation that living near a nuclear power plant does not create any special concerns.[36]

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[32] 1986 Policy Statement on Safety Goals for the Operations of Nuclear Power Plants
[33] Ibid.
[34] Ibid.
[35] Ibid.
[36] Ibid.

While the quantitative risk objectives created a basis for evaluating the risk of nuclear power plant design and operation, they were challenging to implement. Direct measurement of either the early prompt fatalities or long-term cancer fatalities associated with nuclear power plant operation is extremely challenging – bordering on practically infeasible – especially from a regulator's perspective. Demonstrating compliance with these objectives would require extremely detailed analysis and may not have corresponding regulatory value. As a result, the NRC staff developed the following surrogate risk objectives that could be used as more easily calculated risk objectives[37]:

- Core Damage Frequency (CDF)[38] of less than 1×10^{-4}/year
- Large Release Frequency (LRF)[39] of less than 1×10^{-6}/year
- Conditional Containment Failure Probability (CCFP)[40] of less than 0.1

These three objectives were conservatively derived by NRC staff to facilitate use of risk insights to justify operational deviations from the existing prescriptive regulations of conventional large light water reactors. The assumptions underlying these objectives are directly tied to the design, operation, and accident progression of conventional large light water reactors and have significant inherent conservatisms in their calculation.[41] Specifically, these assumptions are tied to the steady state radiological source term of a large light water reactors (2441-megawatt thermal power pressurized water reactor) with a large failure in the reactor pressure boundary that bypasses the reactor building containment. This sequence results in reactor core damage and a large, unmitigated release of radioactive material to the environment.[42] Appendix A provides detailed discussion on the assumptions and calculations that have been used to develop the surrogate risk objectives.

It is important to note that many of the assumptions used in the NRC staff derivation of the surrogate risk objectives, while bounding for large light water reactors, may be excessively conservative for advanced reactors. For example, a microreactor with a much lower thermal power (e.g., 5-megawatt thermal power) would have a steady state radiological source term several orders of magnitude smaller than the source term assumed in the calculation of the

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[37] Public Workshop on Technology Inclusive Risk Metrics for Advanced Reactors
[38] CDF is a metric that describes the likelihood of an event sequence at a nuclear power plant that results in severe damage to nuclear fuel in the reactor and the release of radioactive fission products from the fuel.
[39] LRF is a metric that describes the likelihood that an event sequence at a nuclear power plant that results in severe damage to nuclear fuel in the reactor, the failure or significant bypass of containment barriers, and the release of large quantities of radionuclides to the environment.
[40] CCFP is a metric that describes the likelihood of the failure or significant bypass of containment barriers during a severe fuel damage event sequence that creates a pathway for release of large quantities of radionuclides to the environment.
[41] NUREG-1860, Vol. 2, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing Appendix D"
[42] Evaluation of Severe Accident Risks: Surry Unit 1, NUREG/CR-4551, Vol. 3

existing surrogate risk objectives. Additionally, reactor designs and reactor fuels with improved passive or inherent safety characteristics (e.g., functional containment provided by TRISO fuel particles) could also significantly reduce the release of radiological material during bounding accident sequences.

While the surrogate risk objectives have helped facilitate the more risk-informed and performance-based regulation of existing large light water reactors, the large light water reactor-specific assumptions used to develop the surrogate risk objectives illustrate the inherent limitations of applying the existing surrogate risk objectives to new reactor technologies other than existing large light water reactors. Instead, the development of new technology-inclusive comprehensive risk metrics can be used to support the development of a complete TI-RIPB regulatory framework for advanced reactors.

6.0. Use of new comprehensive risk metrics in TI-RIPB regulation

Development of new comprehensive risk metrics in TI-RIPB regulation requires discussion of the risk and risk limits related to commercial nuclear power plant operation. This section describes a framework to characterize risk regions for nuclear power plants, the current safety limits for nuclear power plants, and the residual risk for nuclear power plant operation that should be evaluated and considered using a comprehensive risk metric.

6.1. Characterizing risk regions for nuclear power plants

The risk posed by nuclear power plant operation can be described based on the quantified risk of different event sequences where the risk of an event sequence is defined as the product of its probability and the consequence of individual events:

$$Risk_{event_i} = Probability_{event_i} \times Consequence_{event_i}$$

The individual event sequences can be visualized based on their probability and consequence using a frequency-consequence (F-C) plot. Figure 1 below shows an F-C plot based on the maximum dose received by an off-site individual and event probability.

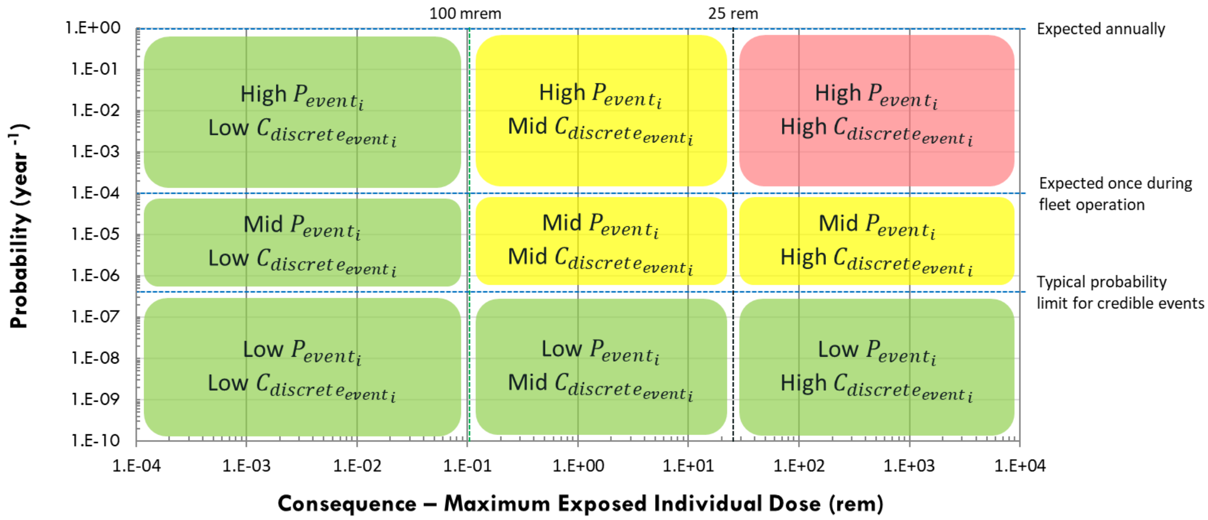

Figure 1. Example F-C Plot for Nuclear Reactor Event Sequences[43]

Different event sequences in Figure 1 can be divided into 9 "risk regions" with similar relative event sequences probability and consequences based on two event sequence probabilities and two event sequence consequences thresholds. A "risk region" is defined as the set of event sequences that share similar qualitative event sequence probabilities and consequences.

The two event sequence probability thresholds used in Figure 1 are:

- Event sequences with mean frequencies of 10^{-4} per year: These event sequences would be expected to occur once during the operational life of a fleet of reactors deployed across multiple sites and operated for decades (e.g., , the probability of an event occurring once across a fleet of 100 reactors operated for 100 years each). This event sequence probability is also the cut-off between the design basis event (DBE) and beyond design basis event (BDBE) sequences defined in the LMP process.[44] Anticipated operational occurrences (AOOs) and DBEs in the LMP process would be grouped together above this sequence probability threshold while BDBEs would be below this threshold.
- Event sequences with mean frequencies of 5×10^{-7} per year: This event sequence probability is the cut-off for events that are considered credible for the design and evaluation of nuclear power plants. Below this probability threshold, the frequency of individual events is considered sufficiently small to be non-credible for safety evaluations. This event sequence probability is the lower threshold frequency for

[43] NIA Perspectives on Advanced Reactor Risk Metrics
[44] NEI 18-04, Rev. 1, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactors"

BDBE event sequences defined in the LMP process.[45] Event sequences with mean frequencies below this threshold are considered sufficiently low probability event sequences that can be screened out as part of the regulatory analysis process.

The two event sequence consequence thresholds used in Figure 1 are:

- 100 mrem offsite dose: typical chronic dose limit for members of the public.[46] This corresponds to event sequences where the consequences would be bounded by regulatory limits associated with public exposure during annual operation of a nuclear power plant.
- 25 rem offsite dose: typical acute dose limit for members of the public[47] This corresponds to event sequences where the consequences would be bounded by the regulatory limits associated with the siting of nuclear power plants to limit public exposure during accidents.

Based on these thresholds, 9 separate "risk regions" can be qualitatively characterized based on both event sequence probability (low, medium, or high) and event sequence consequence (low, medium, or high). Events with low event probability or consequence are qualitatively characterized as low-risk (shown in green in Figure 1), events with high event probability and consequence are qualitatively characterized as high-risk (shown in red in Figure 1), and all other events are characterized as medium-risk (shown in yellow in Figure 1). These different "risk regions" and the qualitative descriptions are defined in Table 1.

Table 1. Risk Region Qualitative Classification

	Low Consequence ($C < 100$ mrem)	**Medium Consequence (100 mrem $< C <$ 25 rem)**	**High Consequence ($C >$ 25 rem)**
High Probability ($F > 10^{-4}$)	Low risk – event sequence consequence will not exceed annual dose limits	Medium risk – event sequence with significant consequences is expected during lifetime fleet operation	High risk – event sequence with severe consequences is expected during lifetime fleet operation
Medium Probability ($10^{-4} > F > 5\times10^{-7}$)	Low risk – event sequence consequence will not exceed annual dose limits	Medium risk – event sequence with significant consequences is highly unlikely but is considered credible during lifetime fleet operation	Medium risk – event sequence with severe consequences is highly unlikely but is considered credible during lifetime fleet operation
Low Probability ($F < 5\times10^{-7}$)	Low risk – event sequence consequence will not exceed annual dose limits and event sequence probability is sufficiently small that event is unlikely to ever occur	Low risk – event sequence probability is sufficiently small that event is unlikely to ever occur	Low risk – event sequence probability is sufficiently small that event is unlikely to ever occur

[45] NEI 18-04, Rev. 1, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-LWRs"
[46] 10 CFR Part 20.1301
[47] 10 CFR Part 100

This qualitative characterization of risk regions into the low, medium, and high risk regions enables the discussion of which risk regions are adequately bounded by existing safety limits and which risk regions may require additional evaluation to understand their contribution to the overall risk of the nuclear power plant. Section 6.2 will describe the existing safety limits for nuclear power plants and Section 6.3 will discuss which specific risk regions may not be bounded by the existing safety limits.

6.2. Characterizing safety limits for nuclear power plants

The safety of commercial nuclear power plants can be generally characterized based on three sets of safety limits: continuous or chronic safety limits, acute safety limits, and cumulative safety limits.

Continuous or chronic safety limits are the limits that apply during normal operation and represent the maximum acceptable impact of operation on an ongoing basis. Continued exposure to hazard levels at or below these chronic safety limits should not have unacceptable cumulative adverse effects. Examples of continuous safety limits include annual public exposure limits (e.g., annual dose limit of 0.1 rem total effective dose equivalent [TEDE] for members of the public[48]), annual worker dose limits (e.g., annual dose limit of 5 rem TEDE for workers[49]), and environmental release limits (e.g., effluent concentration release limits for radionuclides[50])[51].

Acute safety limits are the limits that apply for individual events during plant operation and represent the maximum acceptable impact to an affected group for a single event. Individual exposures to hazard levels at or below these limits during single events should not have unacceptable adverse effects. Examples of acute safety limits include maximum worker exposures during emergencies (e.g., maximum single planned special exposure limit of 25 rem TEDE[52]) and maximum public exposure during emergencies (e.g., reactors may not be sited such that a member of the public receives in excess of 25 rem TEDE during any credible accident[53]). It is important to note that while these acute safety limits are significantly higher than the chronic safety limits, they are still below the level at which acute fatalities will occur from radiation exposure – approximately 500 rem TEDE.[54]

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[48] 10 CFR Part 20.1301
[49] 10 CFR Part 20.1201
[50] 10 CFR Part 20 Appendix B
[51] Annual exposure to natural background radiation is typically quantified as approximately 300 mrem (0.3 rem) and man-made background radiation is typically quantified as approximately 300 mrem (0.3 rem). This results in a typical annual background radiation exposure of approximately 600 mrem (0.6 rem).
[52] 10 CFR Part 20.1206
[53] 10 CFR Part 100
[54] High Radiation Doses NRC Summary

The third set of safety limits reflects the cumulative risk posed by nuclear power plant operations on workers, public, and the environment. This reflects the statistically expected harm from operation over time (or calculated on an annualized basis) and includes the cumulative risk of all possible event sequences. This is the quantitative numerical risk of the nuclear power plant. Nuclear power plants have historically had unique risk profiles compared with other energy sources because the cumulative risk from nuclear power operation was dominated by low-probability, high-consequence events as opposed to high-probability, lower consequence events. Quantitative evaluation of cumulative risk enables explicit consideration of the cumulative impact of all event sequences including low probability, high-consequence events.

In the existing regulatory frameworks, a combination of prescriptive, deterministic, and technology-specific regulatory requirements effectively bound the cumulative risk of existing nuclear power plants. The set of requirements on siting, design, analysis, construction, operation, maintenance, and other activities create a set of parameters that effectively limit the maximum consequence of events, reduce the probability of events, and limit the total number of events. As a result of these overlapping requirements, the calculated cumulative risk for existing power plants licensed with existing LWR-specific regulatory frameworks are relatively similar and meet the Commission's surrogate risk objectives – despite the fact that the plants were designed and built without the formal calculation of the risk surrogates for the design.

Figure 2 shows the calculated CDF for operating pressurized water reactors (PWRs) and boiling water reactors (BWRs). This figure shows the relatively small variation in CDF for the different operating plants despite their initial licensing without an explicit evaluation of a cumulative risk metric.

(internal events at-power for U.S. plants only)

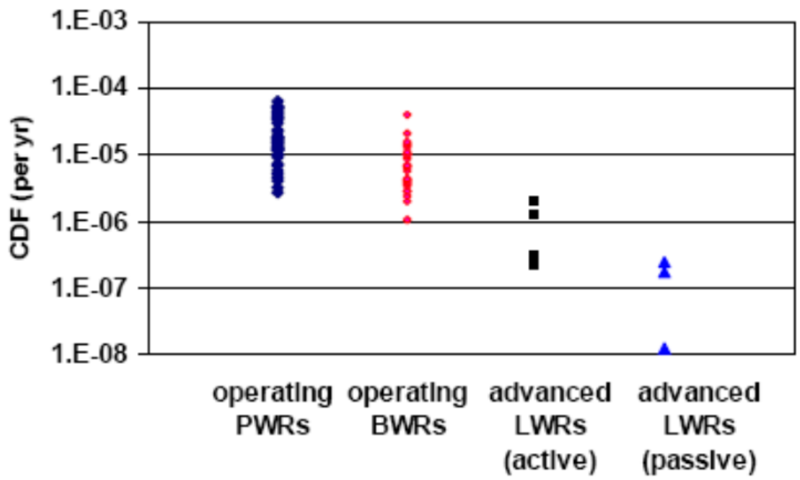

Figure 2. Calculated Core Damage Frequency for Operating Reactors and Proposed LWRs[55]

A complete TI-RIPB regulatory framework would remove many of the prescriptive, deterministic, and technology-specific regulatory requirements on siting, design, analysis, construction, operation, maintenance, and other activities that effectively bounded the overall cumulative risk of nuclear power plants. This gap in the evaluation of plant safety highlights the need for the new third safety limit – a comprehensive risk metric – for reactor licensing in a TI-RIPB regulatory framework that is not necessary in the existing regulatory framework.

6.3. Characterizing comprehensive risk metric needs

An effective comprehensive risk metric should enable evaluation of the overall cumulative risk of a nuclear power plant licensed using a TI-RIPB regulatory framework as part of an assessment of the plant safety basis. Table 2 summarizes the nine different risk regions described in Section 6.1and describes whether the risk region is effectively bounded by the chronic and acute safety limits described in Section 6.2. Table 2 highlights that three risk regions (medium consequence - medium probability events, and medium and high consequence - low probability events) are not effectively bounded by either the chronic safety limits or acute safety limits. Therefore, the cumulative risk associated with these events is not bounded by proposed TI-RIPB regulatory requirements.

[55] Risk Metrics for Operating New Reactors (ML090910608)

Table 2. Risk Region (F-C) Cumulative Characterization and Disposition

<table>
<tr><th></th><th>Low Consequence
(C < 100 mrem)</th><th>Medium Consequence
(100 mrem < C < 25 rem)</th><th>High Consequence
(C > 25 rem)</th></tr>
<tr><td>High Probability
(F > 10^{-4})</td><td rowspan="3">Cumulative risk bounded by chronic safety limits
(dose exposure limits)</td><td>Cumulative risk bounded by chronic and acute safety limits (dose exposure limits)</td><td>Cumulative risk bounded by acute safety limits
(siting limits)</td></tr>
<tr><td>Medium Probability
(10^{-4} > F > 5×10^{-7})</td><td colspan="2" rowspan="2">Cumulative risk not fully bounded by limits</td></tr>
<tr><td>Low Probability
(F < 5×10^{-7})</td></tr>
</table>

Table 2 provides several key insights into the need for a comprehensive risk metric:

- Low consequence events (i.e., events with consequences of less than 100 mrem off-site dose) of any probability are effectively bounded by the chronic safety limits which account for the total exposure of populations to normal operations. If the cumulative effect of these events were to exceed safety limits, they could be managed administratively by enforcing the existing chronic safety limits. A cumulative risk metric is not explicitly needed in these risk regions to help ensure or bound safe operation.
- Medium consequence, high probability events (i.e., events with consequences of between 100 mrem and 25 rem off-site dose and probability of greater than 10^{-4}) are either effectively bounded by the chronic safety limits (based on the assumption of frequent occurrence and enforcement of existing chronic safety limits) or are bounded by the acute safety limits that bound the consequences of analyzed events to less than 25 rem off-site. The overlap of these two limits results in an effective limit on risk because these events are assumed to happen at least once during the operational lifetime of the reactor technology. A cumulative risk metric is not explicitly needed in these risk regions to help ensure or bound safe operation.
- High consequence, high probability events (i.e., events with consequences of greater than 25 rem off-site dose and probability of greater than 10^{-4}) are effectively bounded by the acute safety limits that bound the consequences of analyzed events to less than 25 rem off-site. Enforcement of the acute safety limits would not permit licensing of a plant with credible event sequences (e.g., AOO or DBE within LMP) with high consequences. A cumulative risk metric is not explicitly needed in these risk regions to help ensure or bound safe operation.
- The remaining four regions (medium and high consequence - medium probability events, and medium and high consequence - low probability events) are not effectively bounded by either the chronic safety limits or acute safety limits. These

events have sufficiently low probability that they may not be explicitly evaluated or bounded as part of the licensing basis evaluations for other limits but could still have substantial consequences. While their individual risk may be limited, the cumulative effect of their operation could be substantial and is not effectively bounded by the other limits. While the probability of event sequences in these regions is very low (e.g., BDBE or event sequences below BDBE in the LMP process), the risk contribution from these event sequences could be significant given sufficiently high consequence. The cumulative risk contribution from these regions may also be significant given a sufficiently high number of events. As a result, assessing the individual and cumulative risk contribution from event sequences in these regions is needed to fully characterize or bound a nuclear power plant safety.

The characterization of the risk regions highlights that comprehensive risk metrics can be effectively used to quantitatively assess and bound risk that is otherwise not bounded by chronic or acute safety limits on nuclear power plant operation.

Applicant definition and demonstration of compliance with a comprehensive risk metric enables complete evaluation of the event sequences that contribute to overall risk from nuclear power plants. This will provide the same or greater level of confidence in meeting the NRC legal basis ("reasonable assurance of adequate protection") as the existing regulatory framework using prescriptive, deterministic, and technology-specific regulatory requirements on siting, design, analysis, construction, operation, maintenance, and other activities.

The comprehensive risk metric alone will not provide "reasonable assurance of adequate protection" but can contribute to an overall finding of "reasonable assurance of adequate protection" when assessed with other safety limits and licensing requirements. Use of a comprehensive risk metric as the sole criterion for safety would result in a risk-based regulatory framework and not a risk-informed regulatory framework. A risk-based regulatory framework is extremely challenging to develop and implement due to the inherent uncertainties and operational limitations associated with quantitative risk evaluations.

Demonstration of compliance with a comprehensive risk metric could take several different forms:

- An applicant could perform a quantitative calculation of cumulative risk using licensing basis events and comparing the risk with an existing or proposed comprehensive risk metric or risk surrogate. This would be analogous to an existing LWR applicant demonstrating compliance with the QHOs by completing a full evaluation of the plant CDF, LRF, and CCFP.

- An applicant could develop a hybrid qualitative- and quantitative- evaluation of cumulative risk by selecting and analyzing bounding events and subsequently comparing the calculated risk with an existing or proposed comprehensive risk metric or risk surrogate. This process would be analogous to an existing LWR applicant demonstrating compliance with the QHOs and qualitative health objectives by completing bounding evaluations of the plant CDF, LRF, and CCFP.
- An applicant could develop a qualitative evaluation of cumulative risk by selecting and qualitatively assessing bounding events, comparing the qualitative assessment of cumulative risk of the bounding events with existing standard of "reasonable assurance of adequate protection." This would be analogous to an existing LWR applicant using deterministic and conservative evaluations of the plant to demonstrate "reasonable assurance of adequate protection."

These different approaches to demonstration of compliance with a comprehensive risk metric parallel the different uses of risk insights and deterministic inputs to support risk-informed decision making along a risk-informed continuum within a TI-RIPB regulatory framework.[56]

Creation of an effective framework that enables applicants to propose and demonstrate compliance with a comprehensive risk metric will enable development and implementation of a complete TI-RIPB regulatory framework that is flexible, predictable, self-consistent, and implementable.

6.4. Definition of comprehensive risk metric surrogates

One general challenge with the use of risk metrics for reactor licensing is that it can be difficult to functionally implement the QHOs as a licensing requirement due to the analytic and operational implications of continuous assessment of risk. Calculation of the acute and chronic effects of plant operation requires detailed calculations and may be impractical to use on an ongoing or continuous basis for all plants. Specifically, demonstration of compliance with the QHOs would require the development and maintenance of detailed probabilistic risk assessments (PRAs) that model all events that could result in:

- damage to the reactor core and nuclear fuel (Level 1 PRA),
- release of fission products and other radiological material from damaged fuel to the biosphere (Level 2 PRA), and

[56] Technology-inclusive, Risk-informed, Performance-based Approaches for Development of Licensing Bases Under Part 53

- off-site releases and radioactive contamination and dose exposures to the environment and public (Level 3 PRA).

The challenges of maintaining and using a Level 3 PRA to demonstrate compliance with the QHOs led to the development and usage of the risk surrogates (i.e., CDF, LRF, LERF, CCFP) and incremental changes in these risk surrogates during operation (i.e., ΔCDF, ΔLRF, ΔLERF) for the licensing of existing nuclear power plants.[57]

These risk surrogates can simplify the evaluation and implementation of risk-informed programs for nuclear power plants, but the development of new risk surrogates can require the use of highly conservative assumptions to create bounding risk surrogates applicable to multiple applications or technologies. Characterization of different types of risk surrogates and understanding the underlying assumptions of and relationships between different risk surrogates and the risk metrics is critical to the accurate usage of comprehensive risk metrics and risk surrogates for licensing within an TI-RIPB regulatory framework. Without this type of careful characterization, it is possible to misrepresent or inaccurately compare risk metrics or risk surrogates developed for different applications or different technologies.

Four new categories of risk surrogates are presented below that could facilitate more accurate comparison of different risk surrogates and risk metrics for different applications and reactor technologies:

- Consequence Surrogates
- Release Surrogates
- Harm Surrogates
- Inventory Surrogates

These risk surrogate categories enable more transparent comparison of plant risk and could be used both to ensure appropriate implementation of comprehensive risk metrics for an application and to compare risk metrics across applications and reactor technologies. Table 3 summarizes the different risk surrogates, and the assumptions used compare and evaluate different risk surrogates and risk metrics.

[57] Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement

Table 3. Risk Surrogate Categories

Risk Surrogate Category	Surrogate Description	Example Current Risk Metrics or Risk Surrogate from LWRs
Consequence Surrogate	Surrogate based on the immediate consequences (e.g., dose) or resulting consequences (e.g., cancer and fatalities) from plant event sequences. This surrogate would typically be evaluated using a Level 3 PRA or similar method. This surrogate can be related to other risk metrics and risk surrogates based on the assumptions on the relationship between dose and consequence (e.g., conditional probability of individual cancer or fatality based on radiation doses from a specific release event sequence).	Quantitative Safety Goal: Risk of cancer fatalities that might result from reactor operation should not exceed 0.1% of all other cancer causes[58]
Release Surrogate	Surrogate is based on the magnitude and frequency of releases of hazardous material from plant event sequences. This surrogate would typically be evaluated using a Level 2 PRA or similar method. This surrogate can be related to the Consequence Surrogate based on the event release conditions assumptions (e.g., conditional probability of an individual cancer or fatality for a specific release event sequence).	Large Release Frequency (LRF): 10^{-6} / year[59]
Damage Surrogate	Surrogate is based on the damage event frequency from plant event sequences. This surrogate would typically be evaluated using a Level 1 PRA or similar method. This surrogate can be related to the Release Surrogate based on the event progression assumptions (e.g., conditional probability of a release based on specific damage event sequence).	Core Damage Frequency (CDF): 10^{-4} / year[60]
Maximum Credible Hazard Surrogate	Surrogate is based on the maximum credible release based on plant design and operation. This surrogate would typically be evaluated by quantifying overall plant hazards and comparing them with overall risk goals. This surrogate can be related to the Consequence Surrogate based on assumptions related to the maximum credible releasable inventory, maximum credible release and event sequence conditions, and a bounding assessment of the frequency of credible releases.	Alternative Evaluation for Risk Insights (AERI)[61]: postulated bounding event dose at 100 meters does not exceed dose limits for short-and long-term exposures[62]

[58] NUREG-1860, Vol. 2, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing Appendix D”
[59] Ibid.
[60] Ibid.
[61] Alternative Evaluation for Risk Insights (AERI) was new analysis methodology created to support the NRC staff development of a deterministic regulatory framework within Part 53. The methodology used bounding event analyses and conservative event frequences to enable applicants to demonstrate compliance with QHOs without the development and maintenance of probabilistic risk assessment. The AERI methodology was removed from the proposed Part 53 rule when the Commission voted to eliminate the Part 53 deterministic regulatory framework (Framework B) from the proposed rule in 2024. Future potential uses of the AERI methodology for other new reactor licensing activities is under evaluation by the NRC staff.
[62] Alternative Evaluation for Risk Insights (AERI) Framework - Preliminary Draft Regulatory Guide

7.0. Implementation of comprehensive risk metrics

A new TI-RIPB regulatory framework could help provide a flexible, predictable, self-consistent, and implementable licensing process for advanced reactors and create a more efficient and effective future licensing process. A complete TI-RIPB regulatory framework, however, requires development and implementation of comprehensive risk metrics as an essential regulatory requirement that helps ensure the safe operation of any advanced reactor technology. These comprehensive risk metrics and surrogates must be:

- Flexible: enabling evaluation of any reactor siting, design, analysis, construction, maintenance, and operation
- Predictable: facilitating licensing for different applicants with a clear understanding of regulatory requirements, schedule, and necessary resources
- Self-consistent: providing the same (or greater) level of safety from existing and new nuclear power plants
- Implementable: ensuring that any new regulatory requirements do not create unnecessary burden for applicants or the regulator to demonstrate safety

The comprehensive risk metric alone will not provide "reasonable assurance of adequate protection" but can contribute to an overall finding of "reasonable assurance of adequate protection" when assessed with other safety limits and licensing requirements such as chronic and acute dose limits for members of the public, worker dose limits, environmental effluent limits, and other operational program requirements.

The NRC, industry, and other stakeholders should collaborate on the development of specific comprehensive risk metrics through workshops and other public forums to ensure that future risk metrics are usable by applicants, reviewable by NRC, implementable by operators, and contribute to a complete TI-RIPB licensing basis for new nuclear reactors.

Development of effective comprehensive risk metric requirements in 10 CFR Part 53 for advanced reactors can help create a robust TI-RIPB regulatory process that facilitates more effective "licensing by rule" and reduces applicant reliance on "licensing by order". This change in regulatory framework can enable the more effective and efficient licensing and subsequent deployment of advanced nuclear energy as a climate solution.

Appendix A. Derivation of Commission Quantitative Health Objectives (QHOs) and Risk Surrogate Objectives

This appendix derives the quantitative values currently used by the NRC to assess risk to help stakeholders characterize the underlying assumptions and conservatism in the existing objectives. The numerical values for the NRC quantitative health objectives (QHOs) and the Commission risk surrogate objectives are derived in this appendix.

Commission quantitative health objectives and risk surrogate objectives

The Commission's quantitative health objectives are defined as:[63]

1. The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
2. The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

The Commission's risk surrogate objectives are defined as: [64]

- Core Damage Frequency (CDF) of less than 1×10^{-4}/year
- Large Release Frequency (LRF) of less than 1×10^{-6}/year
- Conditional Containment Failure Probability (CCFP) of less than 0.1

These three risk surrogate objectives were conservatively derived by NRC staff in the late 1980s and early 1990s to support the regulation of conventional large light water reactors. The assumptions underlying these objectives are directly tied to the design, operation, and accident progression of conventional large light water reactors and have significant inherent conservatisms in their calculation.

Calculation of quantitative risk objectives

The quantitative risk objectives can be calculated based on Commission policy using measurements of other societal risk and cancer fatalities.

[63] 1986 Policy Statement on Safety Goals for the Operations of Nuclear Power Plants
[64] Risk Metrics for Operating New Reactors

Calculation of prompt fatality risk objective

The quantitative risk objective for prompt fatalities is calculated using the following equation:

$$IER = 0.001 \cdot ISR$$

where:

- IER is the individual acute risk objective, and
- ISR is the individual risk from all other sources of risk in society.

The NRC has historically used an ISR of 5×10^{-4} fatalities per year, which results in a calculated IER of 5×10^{-7} fatalities per year.[65]

Selection of an ISR is challenging due to the many societal factors that can affect the annual accident rate and the fact that these numbers will change over time. Review of prior NRC research on risk provides insights into the accidents that were included in the assumed ISR of 5×10^{-4} fatalities per year.[66] It is important to note that these numbers were originally tied to the expected leading causes of death in the mid-1970s and were dominated by motor vehicle accident fatalities (approximately 50% of the assumed ISR of 5×10^{-4} fatalities per year[67]). The ISR will change annually based on changing societal risks and behavior.

Review of updated accident data from the U.S. National Safety Council provides insights on the current calculated ISR in the United States for accidents to which members of the U.S. population are generally exposed. The United States has seen a significant rise in the ISR since 2016, increasing from 5×10^{-4} fatalities per year from accidental injuries in 2016 to 6.8×10^{-4} fatalities per year from accidental injuries in 2022.[68] These increases, however, were largely driven by significant increases in the accidental deaths related to drug overdoses, accounting for an increase between 2016 and 2022 of 1.3×10^{-4} fatalities per year.

[65] NUREG-1860, Vol. 2, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing Appendix D”
[66] NUREG/CR-1916, "A Risk Comparison."
[67] NUREG/CR-1916, "A Risk Comparison."
[68] Odds of Dying - Data Details - Injury Facts

Table A1 summarizes and compares major components of the observed ISR in 1977, 2016, and 2022, and Figure A1 illustrates the annual ISR since 1903 in the United States.

Table A1. Selected ISR Annual Accident Fatality Rates[69,70]

	2022	2016	1977
Total Accidental	6.81E-04	4.99E-04	4.8E-04
Motor Vehicle Accidents	1.38E-04	1.25E-04	2.3E-04
Falls	1.40E-04	1.07E-04	0.6E-04
Drug Overdoses	2.99E-04	1.69E-04	0.2E-04
All Other Causes	1.04E-04	9.77E-05	1.7E-04

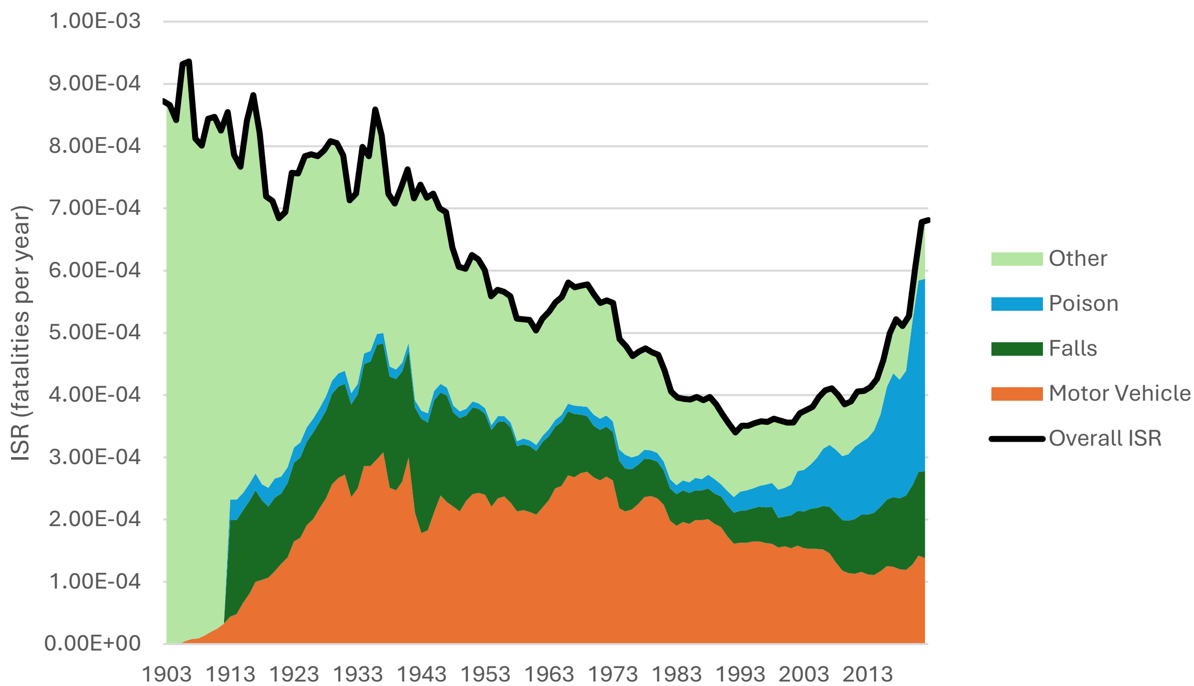

Figure A1. ISR Annual Accident Fatality Rates[71]

Use of an ISR of 5×10^{-4} to 5.5×10^{-4} for the calculation of the IER appears reasonable but depends on assumptions about which accidents to which members of the U.S. population are generally exposed. Inclusion of accidental deaths related to drug overdoses in recent years would result in a higher observed ISR and produced a higher IER in calculations of 6.8×10^{-7} fatalities per year. Including the accident deaths associated with

[69] NUREG/CR-1916, "A Risk Comparison."
[70] Odds of Dying - Data Details - Injury Facts
[71] Historical Trends: Deaths by Cause - Injury Facts

drug overdoses, however, may not be appropriate because they may not be classified as accidents "to which members of the U.S. population are generally exposed".

Calculation of latent fatality risk objective

The quantitative risk objective for latent (long-term) cancer fatalities resulting from release of radiation from nuclear power plant accidents is calculated using the following equation:

$$ILR = 0.001 \cdot ICR$$

where:

- ILR is the individual latent cancer risk[72] objective, and
- ICR is the individual cancer fatality risk from all other sources.

The NRC has historically used an ICR of 2×10^{-3} cancer fatalities per year and results in a calculated ILR of 2×10^{-6} cancer fatalities per year.[73]

Selection of an ICR is challenging due to the many societal factors that can affect the annual cancer fatality rate and that these numbers will change over time based on factors including cancer incidence related to carcinogen exposure, overall population health, and evolving treatments for cancer that can lower mortality rates. Prior NRC research on risk provides a basis for the assumed ICR of 2×10^{-3} cancer fatalities per year based on an annual cancer fatality rate of 1.8×10^{-3} cancer fatalities per year.[74]

Review of updated cancer data from the U.S. National Safety Council provides insights on the current calculated ICR in the United States for cancer fatality risks resulting from all other causes. Overall, the United States has seen a continually decreasing ICR since 1999. The ICR decreased from 2×10^{-3} in 1999 to 1.4×10^{-3} in 2022.[75]

The observed changes in ICR over time, however, differ significantly for different population groups includes by gender, by race and ethnicity, and by location. Table A2 summarized the ICR for these different groups in 2022. Figures A2, A3, and A4 illustrate the annual ICR since 2000 in the United States by gender (Figure A2), race (Figure A3), and by location (Figure A4 for the U.S. states with the lowest and highest ICR).

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[72] Individual latent cancer risk describes the annual individual probability of a latent cancer-related death due to exposure to ionizing radiation. The term "latent" is used to characterize health effects that may occur years or decades after exposure.
[73] NUREG-1860, Vol. 2, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing Appendix D"
[74] NUREG/CR-1916, "A Risk Comparison"
[75] US HHS SEER Cancer Data Explorer

Table A2. Selected 2022 ICR Cancer Mortality Rates

Characteristic	Populations	Fatality Rate per Year
Location	Mississippi	1.78E-03
Gender	Male	1.67E-03
Race/Ethnicity	Black	1.63E-03
Race/Ethnicity	American Indian / Alaska Native	1.55E-03
Race/Ethnicity	White	1.48E-03
United States Average		1.42E-03
Gender	Female	1.24E-03
Location	Utah	1.16E-03
Race/Ethnicity	Hispanic	1.05E-03
Race/Ethnicity	Asian / Pacific Islander	9.15E-04

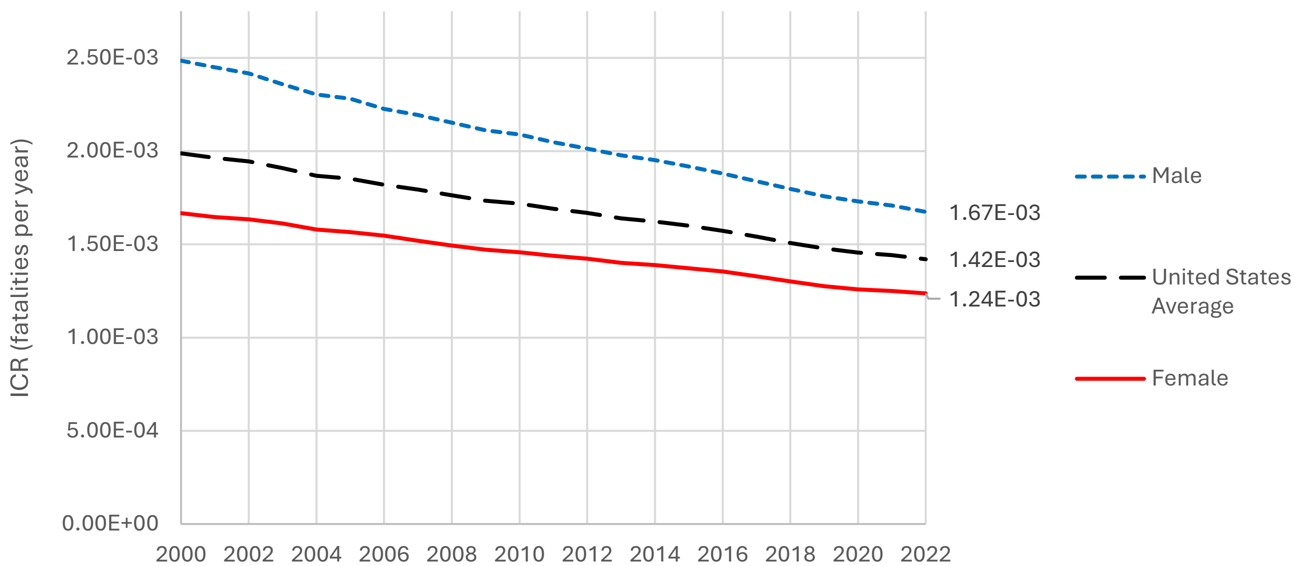

Figure A2. ICR Annual Cancer Fatality Rates by Gender[76]

[76] Historical Trends: Deaths by Cause - Injury Facts

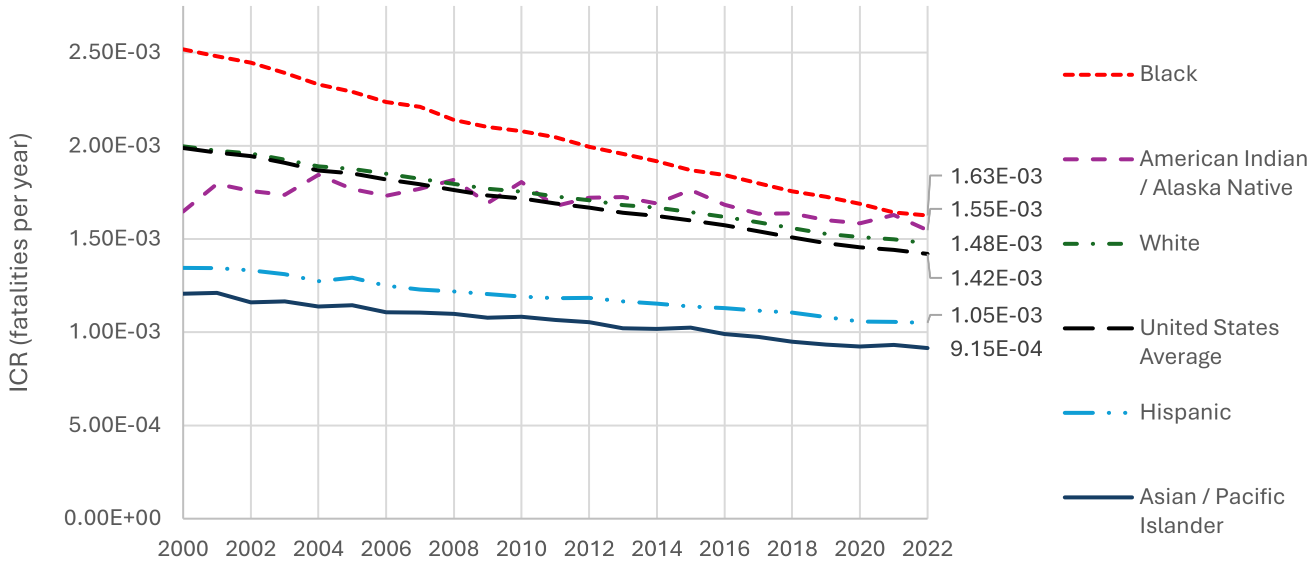

Figure A3. ICR Annual Cancer Fatality Rates by Race and Ethnicity[77]

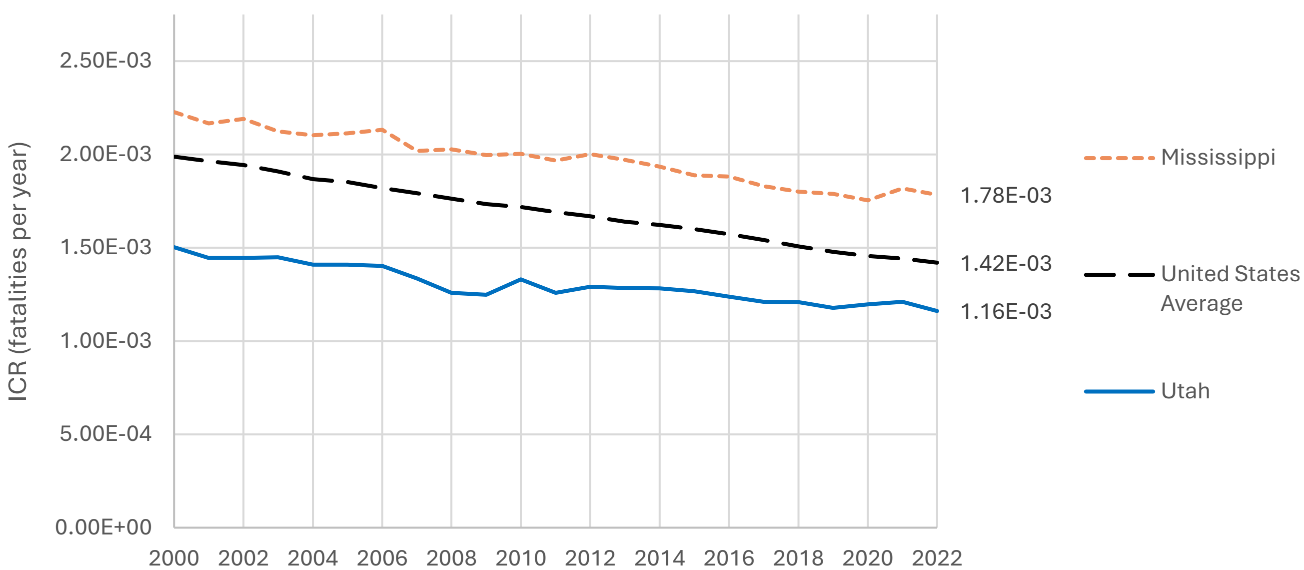

Figure A4. ICR Annual Cancer Fatality Rates by Location[78]

Use of an ICR of 1.5×10^{-3} to 2×10^{-3} for the calculation of the ILR appears reasonable because it conservatively bounds all population groups but depends on the specific assumptions about cancer fatality risks resulting from all other causes in different populations. Use of an ICR of 1.4×10^{-3} based on the current national average ICR for the

[77] Historical Trends: Deaths by Cause - Injury Facts
[78] Ibid.

calculation of the ILR may not be conservative because several key groups have an ICR significantly lower than the national average. For example, use of an ICR of 1.4×10^{-3} would result in an ILR of greater than one-tenth of one percent of the individual cancer fatality risk for women, Hispanic, and Asian/Pacific Islander populations.

Derivation of surrogate risk metrics

The generally acceptable risk surrogates used by the NRC can be derived based on NRC staff calculations in NUREG-1860[79]. The two main acceptable risk surrogates used by the NRC used to support regulatory decision making are:

- Large Early Release Frequency (LERF) of less than 1×10^{-5}/year
- Core Damage Frequency (CDF) of less than 1×10^{-4}/year

It is important to note that while the NRC Commission initially used the large release frequency (LRF) as a risk surrogate for plant safety in the 1986 Policy Statement[80], the NRC and applicants now generally use the large early release frequency (LERF) as the risk surrogate for plant safety with respect to acute radiation exposure during nuclear accidents. Event sequences encompassed by the LERF occur quickly and do not provide time for off-site emergency response activities (e.g., evacuations) that could mitigate the effects of a large radiological release on the public. Therefore, the LERF risk surrogate more closely aligns with a risk metric that characterizes near-term, acute risk from nuclear accidents.

The CDF and LERF risk surrogates are rederived below to highlight the technology- and design-specific assumptions used to calculate the risk surrogates based on the quantitative risk metrics. These risk surrogates were developed by NRC staff using the results from the Surry Nuclear Power Station Level 3 PRA.[81]

Derivation of large early risk frequency risk surrogate

The LERF risk surrogate for prompt fatalities is calculated based on the maximum individual acute risk objective of IER less than 5×10^{-7}. The LERF risk surrogate is derived by NRC staff in NUREG-1860 and is shown below.[82]

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[79] NUREG-1860, Vol. 2, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing Appendix D"

[80] 1986 Policy Statement on Safety Goals for the Operations of Nuclear Power Plants

[81] NUREG-1860, Vol. 2, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing Appendix D"

[82] Ibid.

The total individual acute risk of an early fatality can be calculated as the sum of the risk of all event sequences that result in acute fatalities:

$$IER = \sum_{i=1}^{N} \frac{EF_i \cdot LERF_i}{P_i}$$

where:

- N is the number of event sequences that result in acute fatalities
- EF_i is the number of acute fatalities resulting from event sequence i
- $LERF_i$ is the probability of a large early release that creates event sequence i
- P_i is the population exposed to the event sequence i

The number of acute fatalities resulting from event sequence i (EF_i) can be calculated as:

$$EF_i = P_i \cdot CPEF_i$$

where:

- $CPEF_i$ is the conditional probability of an individual becoming a prompt (or early) fatality for event sequence i

Combining these two equations allows for evaluation of IER independent of evaluated population:

$$IER = \sum_{i=1}^{N} CPEF_i \cdot LERF_i$$

Based on the maximum individual acute risk objective of IER less than 5×10^{-7}, the following assumptions are made by NRC staff to determine an acceptable LERF:

- One event sequence (n) dominates the maximum individual acute fatality risk
- The dominating event sequence is a worst-case release involving an early and large failure of containment that leads to a large, unmitigated release before effective evacuation of surrounding populations can occur

Using these assumptions, the bounding LERF risk surrogate is calculated using the worst case $CPEF_n$ from the Surry Nuclear Power Station Level 3 PRA[83]:

$$LERF = \frac{IER}{CPEF_n}$$

where:

- $CPEF_n$ is the maximum conditional probability of an individual becoming a prompt (or early) fatality (CPEF) for event sequence n

The maximum conditional probability from the Surry Nuclear Power Station Level 3 PRA used in the NRC LERF calculation is a $CPEF_n$ of 3×10^{-2}. Using the maximum individual acute risk objective IER of 5×10^{-7}, the corresponding maximum LERF risk surrogate value is 1.67×10^{-5}. This value was rounded down for conservatism and a LERF risk surrogate value of less than 1×10^{-5}/year is used as the benchmark for plant evaluations.

Review of this risk surrogate calculation reveals significant conservatisms and challenging assumptions. The NRC risk surrogate calculation assumes that a single event sequence (n) dominates the maximum individual acute fatality risk and that the cumulative IER evaluation can be approximated by a single term. Approximation of the summation interval using a limited number of terms is desirable because it enables simplified approximation of an interval that may contain a large number of different low-significance event sequences. One challenge with this approximation, however, is that a single event sequence may not dominate the maximum individual acute fatality risk.

Table A.3. summarizes the results from the Surry Nuclear Power Station Level 3 PRA that were used to calculate the NRC LERF risk surrogate. The table provides the CPEF, LERF, and sequence IER for each source term. Additionally, the table lists the contribution of the individual source term to the total IER for the Surry Level 3 PRA.

Review of the Level 3 PRA results shows that while the bounding source term used to calculate the LERF surrogate (SUR-10-3) has the highest CPEF (3×10^{-2}) of any early release source term (indicated by SUR-XX-3 source term identifier), it is not the highest CPEF of the source terms analyzed and was not the dominant contributor to plant IER. This shows that while use of the SUR-10-3 CPEF is a conservative assumption for the calculation of the LERF using the total IER, it is extremely conservative and does not reflect the realistic LERF associated with the specific source term and event sequence.

[83] Evaluation of Severe Accident Risks: Surry Unit 1, NUREG/CR-4551, Vol. 3

This derivation also highlights that the calculation is based solely on the data from the Surry Level 3 PRA. Calculation of the LERF using plant-specific data would likely result in different, plant-specific LERF values that better align with the actual risk characteristics of the individual plant.

Table A3. Evaluated Early Individual Risk Contributors for Surry Level 3 PRA[84]

Analysis Source Term	CPEF	LERF	Sequence IER	% of Total IER
SUR-07-2	3.4E-02	1.1E-07	3.8E-09	24%
SUR-03-3	4.4E-03	7.3E-07	3.2E-09	20%
SUR-11-3	2.2E-02	1.2E-07	2.8E-09	17%
SUR-05-3	1.6E-02	9.4E-08	1.5E-09	10%
SUR-10-3	2.9E-02	4.5E-08	1.3E-09	8%
SUR-07-3	9.0E-03	1.3E-07	1.2E-09	8%
SUR-06-3	1.3E-02	6.9E-08	9.3E-10	6%
All Other Events			1.2E-09	7%

Derivation of core damage frequency risk surrogate

The CDF risk surrogate for latent fatalities is calculated based on the maximum individual latent cancer risk objective of ILR less than 2×10^{-6}. The CDF risk surrogate is derived by NRC staff in NUREG-1860 and is shown below.[85]

The total individual risk of latent cancer fatalities can be calculated as the sum of the risk of all event sequences that result in population dose exposure:

$$ILR = \sum_{j=1}^{M} \frac{LF_j \cdot LLRF_j}{TP_{10}}$$

where:

- M is the number of event sequences that result in population dose exposure
- LF_j is the number of latent cancer fatalities resulting from event sequence j
- $LLRF_j$ is the probability of a release that creates event sequence j
- TP_{10} is the population exposed within 10 miles of the reactor

[84] Evaluation of Severe Accident Risks: Surry Unit 1, NUREG/CR-4551, Vol. 3

[85] NUREG-1860, Vol. 2, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing Appendix D”

The number of latent cancer fatalities resulting from event sequence j (LF_j) can be calculated as:

$$LF_j = TP_{10} \cdot CPLF_j$$

where:

- $CPLF_j$ is the conditional probability of an individual becoming a latent cancer fatality for event sequence j

Combining these two equations allows for evaluation of ILR independent of evaluated population:

$$ILR = \sum_{j=1}^{M} CPLF_j \cdot LLRF_j$$

Based on the maximum individual acute risk objective of ILR less than 2×10^{-6}, the following assumptions are made by NRC staff to determine an acceptable CDF:

- One event sequence (m) dominates the maximum individual acute fatality risk
- The dominating event sequence is a worst-case release involving a large failure of containment that leads to a large, unmitigated release that occurs after effective evacuation of surrounding populations and only produces latent health effects

The probability of a release that creates event sequence j ($LLRF_j$) can be calculated using the above assumptions:

$$LLRF_M = CDF_M \cdot CPLLR_M$$

where:

- CDF_M is the core damage frequency for the bounding event sequence that results in population dose
- $CPLLR_M$ is the conditional probability of a large release that results in population dose. The assumption of a worst-case release that results in failure of containment and unmitigated release results in a $CPLLR_M$ of 1.

Using these assumptions, the bounding CDF risk surrogate is calculated using the worst case $CPEF_n$ from the Surry Nuclear Power Station Level 3 PRA[86]:

$$CDF = \frac{ILR}{CPLF_M}$$

where:

- $CPLF_M$ is the maximum conditional probability of an individual becoming a latent cancer fatality for bounding event sequence M

The maximum conditional probability in the Surry Nuclear Power Station Level 3 PRA is a $CPLF_M$ of 4×10^{-3}. Using the maximum individual acute risk objective ILR of 2×10^{-6}, the corresponding maximum CDF risk surrogate value is 5×10^{-4}. This value was rounded down for conservatism (to "generally assure that the latent cancer QHO is met") and a CDF risk surrogate value of less than 1×10^{-4}/year is used as the benchmark for plant evaluations.

Review of this risk surrogate calculation reveals significant conservatisms and challenging assumptions. The NRC risk surrogate calculation assumes that a single event sequence (m) dominates the maximum individual latent cancer fatality risk and that the cumulative ILR evaluation can be approximated by a single term. Again, approximation of the summation interval using a limited number of terms is desirable because it enables simplified approximation of an interval that may contain a large number of different low-significance event sequences. One challenge with this approximation, however, is that a single event sequence may not dominate the maximum individual acute fatality risk.

Table A.4. summarizes the results from the Surry Nuclear Power Station Level 3 PRA that were used to calculate the NRC CDF risk surrogate. Review of the Level 3 PRA reveals that while the bounding event used in the evaluation (SUR-05-2) has the highest CPLF (4×10^{-3}) of all analyzed event, it is not the highest CPLF of the events analyzed and is not the significant contributor to plant ILR, contributing to less than 0.2% of the plant ILR. Similar to the derivation of the LERF, while the use of the SUR-05-2 CPLF is a conservative assumption for the calculation of the CDF using the total ILR, it is extremely conservative and does not reflect the realistic CDF associated with the specific source term and event sequence.

This derivation also highlights that the calculation is based solely on the data from the Surry Level 3 PRA. Calculation of the CDF using plant-specific data would likely result in different, plant specific CDF values that better align with the actual risk characteristics of the individual plant.

[86] Evaluation of Severe Accident Risks: Surry Unit 1, NUREG/CR-4551, Vol. 3

Table A4. Evaluated Latent Individual Risk Contributors for Surry Level 3 PRA[87]

Analysis Source Term	CPLF	CDF	Sequence ILR	% of Total ILR
SUR-07-2	2.7E-03	1.1E-07	3.1E-10	19%
SUR-03-3	3.6E-04	7.3E-07	2.6E-10	16%
SUR-11-3	7.6E-05	3.2E-06	2.4E-10	15%
SUR-05-3	1.5E-03	1.2E-07	1.9E-10	11%
SUR-10-3	8.9E-04	9.4E-08	8.4E-11	5%
SUR-07-3	6.0E-04	1.3E-07	8.0E-11	5%
SUR-06-3	9.7E-04	6.9E-08	6.7E-11	4%
SUR-05-2	4.0E-03	6.7E-10	2.7E-12	0.2%
All Other Events			1.7E-09	25%

[87] Evaluation of Severe Accident Risks: Surry Unit 1, NUREG/CR-4551, Vol. 3